NUSCALE POWER, LLC

SAFETY EVALUATION FOR NUSCALE TOPICAL REPORT, TR-0915-17772, REVISION 3, "METHODOLOGY FOR ESTABLISHING THE TECHNICAL BASIS FOR PLUME EXPOSURE EMERGENCY PLANNING ZONES AT NUSCALE SMALL MODULAR REACTOR PLANT SITES"

1.0 Introduction

By letter dated June 10, 2022, NuScale Power, LLC (NuScale), submitted Revision 3 of licensing Topical Report (TR) TR-0915-17772 (Reference 1), titled "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor (SMR) Plant Sites." Revision 3 of the TR revises the methodology in Revision 2 (Reference 2). The United States (U.S.) Nuclear Regulatory Commission (NRC) staff issued Requests for Additional Information (RAIs) 9828 and 9830 based on its review of Revision 2 of the TR. In addition to revising the TR to Revision 3, NuScale submitted its responses to RAI 9828 (Reference 3) and RAI 9830 (Reference 4) and incorporated the responses in Revision 3 of the TR.

In Section 2.5.1 of Revision 3 of the TR, NuScale stated that the TR is applicable to only a NuScale SMR design, including the standard plant design (Docket 52-048), and variations and derivatives thereof comprising all the following characteristics:

1. Small modular integral pressurized LWRs, meaning reactor modules composed of a reactor core, primary cooling loop, pressurizer, and steam generators(s) within a reactor vessel, housed within a containment vessel normally operated at sub-atmospheric pressure conditions,

2. Operating modules partially immersed in water that serves as the [ultimate heat sink] UHS,

3. The UHS retained below grade in a structure with up to 12 reactor modules per UHS,

4. A safe shutdown earthquake with a peak ground acceleration of 0.5g, and

5. Structures, systems, and components (SSCs) capable of performing their safety functions without [alternating current] AC, [direct current] DC electric power, or operator actions for at least 72 hours following a design basis event.

Also, in Revision 3 of the TR, NuScale summarized that: (1) the NuScale methodology for establishing the technical basis for plume exposure EPZ sizing considers source terms and dose consequences; (2) the methodology, when implemented with design information as part of an application, provides a basis for sizing the plume exposure EPZ; (3) the methodology is applicable to any EPZ size, including the site boundary; and (4) the final EPZ size that results from applying the methodology is the smallest distance at which the dose consequences of all screened-in accident sequences are less than their respective dose criterion. NuScale further stated that, based on the results of applying the methodology, the final EPZ size may differ from the current 10-mile requirement.

NuScale stated in its TR that “The use of this methodology to determine final EPZ size will occur when an application is submitted to the NRC to construct and operate an advanced reactor design. The most likely mechanism is a combined license (COL) application; however, it is acknowledged that other regulatory processes exist.” This TR applies to Title 10 of *Code of Federal Regulations* (10 CFR) Part 50 (Reference 5) and 10 CFR Part 52 (Reference 6 for rule language; Reference 7 for Statement of Considerations for the 2007 version of Part 52) applicants; however, the TR does not apply to other regulatory processes. The NRC staff has written a “Condition of Use” in Section of 5.0 of this safety evaluation (SE) regarding the use of this TR by a Part 50 applicant. As stipulated in Section 5.0 of this SE (Condition A), the use of this TR by an applicant for an operating license under 10 CFR Part 50 is permitted if:

1. The applicant submits the design- and site-specific PRA results and insights for staff review as part of their Final Safety Analysis Report, consistent with the Commission expectations for the use of the PRA, as described in the Statement of Considerations for 10 CFR Part 52 (72 FR 49387), and

2. The level of design detail in the design- and site-specific PRAs is commensurate with a 10 CFR Part 52 combined license (COL) application.

NuScale also requested that the NRC provide an SE on the design-specific, plume exposure EPZ sizing methodology, including the following: (1) a conclusion that the NuScale proposed plume exposure EPZ sizing methodology in the topical report, when supported by design-specific information and appropriately implemented by each COL applicant, is an acceptable approach for justifying the plume exposure EPZ size for the NuScale design; and (2) identification of any issues related to the technical basis for the NuScale EPZ sizing methodology that are to be resolved prior to, or as part of, the COL review process. NuScale further stated that with an NRC approved EPZ TR, NuScale and each COL applicant would collaborate to develop emergency preparedness elements with each contributing organization having distinct roles and responsibilities, and the COL applicants would apply the approved methodology to their specific sites to determine a plume exposure EPZ.

The NuScale TR does not exclude other applicable emergency planning requirements, for example, those pursuant to 10 CFR 50.47, “Emergency Plans,” (Reference 8) and 10 CFR Part 50 Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,”(Reference 9) which codify the emergency planning requirements for nuclear power reactors.

Section 2.0 of this SE discusses the applicable regulations, Commission policies, and NRC staff guidance relevant to the NRC staff’s review of the TR. Section 3.0 documents the NRC staff’s evaluation of the TR, and Section 4.0 provides the NRC staff’s conclusion on the acceptability of the TR for use by a COL applicant or construction license holder. Section 5.0 provides the Conditions of Use of the TR.

2.0 Regulatory Basis

10 CFR 50.47, “Emergency Plans,” and 10 CFR Part 50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," codifies the emergency planning requirements for nuclear power reactors. These regulations require the plume exposure

pathway EPZ to consist of an area about 10 miles (16 km) in radius for water-cooled reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for gas-cooled reactors and reactors with an authorized power level less than 250 MWt. If a water-cooled reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius.

10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," (Reference 10) refer to various dose-based criteria and limits in terms of total effective dose equivalent (TEDE). The TEDE methodology is based on dosimetry methodologies defined by the International Commission on Radiological Protection (ICRP) in Publication 26, *Recommendations of the ICRP,* and Publication 30, *Limits for Intakes of Radionuclides by Workers*. The dosimetry methodologies are applied in:

> 10 CFR Part 50 – through the TEDE criteria (defined in §50.2) for the design, construction, and operation of the facility under normal and accident conditions. The TEDE is defined, in part, as "… the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures)."
> 10 CFR Part 20 – through the TEDE limits (defined in §20.1003) to establish standards and practices for radiation protection purposes for occupational and public health during normal operation. 10 CFR Part 20, Appendix B, *Appendix B to Part 20—Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage*, provides direction in how to determine external and internal exposures. Among other information, the appendix provides an appropriate method to derive the Annual Limits on Intake and Derived Air Concentrations based on ICRP Publication 30 tissue weighting factors. The tissue weighing factors are directly codified by §20.1003, *Definitions*, within the table labeled, *Organ Dose Weighting Factors*, as follows:

Organ or Tissue	W_T
Gonads	0.25
Breast	0.15
Red Bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bong surfaces	0.03
Remainder	0.3
Whole Body	1

The regulations in 10 CFR 52.79(a)(46) (Reference 11) state that a COL application must contain a Final Safety Analysis Report (FSAR) that includes a description of the plant-specific PRA and its results. The Statement of Considerations for the 2007 version of 10 CFR Part 52 states the understanding that the complete PRA (e.g., codes) would be available for NRC inspection at the applicant's offices, if needed.

The regulations in 10 CFR, 52.79(c)(1), 10 CFR 52.79(d)(1), and 10 CFR 52.79(e)(1) (Reference 11) state that if a COL application references a standard design approval, standard design certification (DC), or the use of one or more manufactured nuclear power reactors licensed under Subpart F of 10 CFR Part 52, then the plant-specific PRA information must use the PRA information for the design approval, DC, or manufactured reactor, respectively, and must be updated to account for site-specific design information and any design changes or departures.

The Statement of Considerations for the 2007 revision of 10 CFR Part 52 states in the case where a COL application is referencing a DC, the NRC only expects the design changes and differences in the modeling (or its uses) pertinent to the PRA information to be addressed to meet the submittal requirement of 10 CFR 52.79(d)(1).

Commission Policy Statements

The NRC/Environmental Protection Agency (EPA) Task Force Report on Emergency Planning, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396 (Reference 12), provides a planning basis for offsite emergency preparedness efforts considered necessary and prudent for large power reactor facilities. The Commission's Policy Statement, "Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents," (Reference 13) directs the NRC staff to incorporate the guidance in NUREG-0396 into emergency preparedness documents. NUREG-0396 is also reflected in the NRC Final Rule on Emergency Planning dated August 19, 1980. The Federal Emergency Management Agency (FEMA) has also concluded that the guidance in NUREG-0396 should be used as the planning basis for emergency preparedness around nuclear power facilities. NUREG-0396 is the technical basis for: (1) the current EPZ regulations for operating power reactors referenced in 10 CFR 50.47; (2) the methodology for draft final Emergency Planning Rule for SMRs and Other New Technologies - 10 CFR 50.160; and (3) this TR.

The Commission Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," (Reference 14), outlines the need for new reactor applicants to complete a Probabilistic Risk Assessment (PRA) and consider severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety.

In August 1995, the NRC adopted the Policy Statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory," (PRA Policy Statement; Reference 15) regarding the expanded use of PRA, which states:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical, within the bounds of the state-of-the art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and NRC staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with, unless these rules and regulations are revised.

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

Commission Direction – Staff Requirements Memorandum

The Staff Requirement Memorandum (SRM) to SECY 90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationships to Current Regulatory Requirements," (Reference 16) states that the Commission supports the use of 1E-4 per year of reactor operation as a core damage frequency goal. Consistent with the Commission's decision on SECY-89-102, the Commission approved the overall mean frequency of a large release of radioactive material to the environment from a reactor accident as less than one in one million per year of reactor operation.

SRM to SECY-04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," (Reference 17) states that "the licensee's submittal is expected to be in conformance with the published standards."

SRM to SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation," (Reference 18) defines the terms and Commission expectations for risk-informed and performance-based regulation. It defines, among other terms, a risk-informed approach to regulatory decision-making as one that "represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety."

Relevant Guidance

The relevant guidance used to support this review is as follows:

NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," provides a basis for Federal, State, and local government emergency preparedness organizations to determine the appropriate degree of emergency response planning efforts in the environs of nuclear power plants. Specifically, the technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396 and was based upon evaluation of the offsite consequences of both design-basis and beyond design basis accidents (BDBA), and a comparison of doses to the EPA guidance entitled, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" (EPA-520/1-75-001, Reference 19), issued September 1975. In January 2017, the EPA published an update, entitled "PAG Manual Protective Action Guides and Planning Guidance for Radiological Incidents" (EPA-400/R-17/001, Reference 20), herein referred to as the "EPA PAG [Protective Action Guide] Manual," which supersedes the previous versions of the manual.

The EPA PAG Manual assists public officials in planning for emergency response to radiological incidents. For purposes of this document, a radiological incident is an event or a series of events, deliberate or accidental, leading to the release or potential release into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions. The PAG Manual provides radiological protection criteria for application to all incidents that would require consideration of protective actions. The EPA emergency response actions include sheltering and evacuation, as given in the PAGs, or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ.

The EPA PAG Manual Section 1.4, "Radiological Incident Phases and Applicability of Protective Actions," discusses the "phases" in which emergency planners divide responses to radiological incidents. The Early Phase PAG is defined as:

> The beginning of a radiological incident for which immediate decisions for effective use of protective actions are required and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions. When available, predictions of radiological conditions in the environment based on the condition of the source or actual environmental measurements may be used. Protective actions based on the PAGs may be preceded by precautionary actions during the period. This phase may last from hours to days.

The EPA PAG Manual, Table 1-1, "Summary Table for PAGs, Guidelines, and Planning Guidance for Radiological Incidents," presents the principal protective actions association with the Early Phase PAG and also the related guidelines, dose-related criteria, and planning guidance. For the Early Phase PAG, the established dose criteria are a range from 1 to 5 rem total effective dose (TED). The EPA TED is computed using dosimetry from the ICRP Publication 60 series, published in 1991. The NRC staff notes that the EPA defined TED is different than the NRC defined TEDE, which is defined in regulation (see Regulatory Basis above), as it utilizes different dosimetry methodologies. As such, the NRC uses its definition of TEDE for regulatory activities under its statutory authority.

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 21) states that "all plant operating modes and hazard groups be addressed when those risk contributions affect the decision."

RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 22) provides an approach for determining whether the base PRA, in total or the parts that are used to support an application, is acceptable for use in regulatory decision making for LWRs. RG 1.200, Revision 3, endorses the joint American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) PRA Standard, ASME/ANS RA-Sa- 2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," which addresses core damage frequency (CDF) and large early release frequency (LERF) for internal and external hazard groups during at-power operations. RG 1.200 includes a summary of technical characteristics and attributes of a Level 1 and Level 2 PRA for low power and shutdown conditions.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed decision-making," (Reference 23) provides guidance for the treatment of uncertainties in a risk-informed application.

Design Certification/Combined License, (DC/COL)-ISG-028 "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application" (Reference 24) provides guidance for the NRC staff review of PRAs supporting design certification and COL applications. This guidance applies to all hazard PRAs, except seismic PRAs. Regarding risk informed applications, the guidance states that, "The positions presented in this guidance should not be relied on to address other types of applications that use PRA results and insights (e.g., risk-informed inservice inspections) or to address PRA requirements for COL holders/licensees (e.g., 10 CFR 50.71(h)(1)). Such applications need to directly address the application-specific regulations and guidance, including the evaluation of the technical adequacy of the PRA needed for the specific application using the PRA Standard, as endorsed by RG 1.200."

3.0 <u>Technical Evaluation</u>

3.1 Method of NRC Staff Review

The NRC staff's review of the TR focused on: (1) the dose criteria, source term development, and dose evaluation methodology; (2) the use of PRA information for EPZ sizing, including the screening of accident sequences for consideration in EPZ sizing; and (3) consistency of the methodology with risk-informed decision-making. This use of risk insights and the screening of accidents sequences was evaluated against NUREG-0396.

NUREG-0396, as discussed in Section E of the report, used risk insights from The Reactor Safety Study (RSS; also known as WASH 1400; Reference 25) for emergency planning considerations. NUREG-0396, Figure I-11 shows the Conditional Probability of Exceeding Whole-Body Dose Versus Distance which includes Reactor Safety Study (RSS) accident release categories for both Pressurized- and Boiling Water Reactor beyond design basis severe accidents. NUREG-0396, page 5, states:

> As an alternative to attempting to define a specific accident sequence, the Task Force decided to identify the bounds of the parameters for which planning is recommended based upon a knowledge of the potential consequences, timing, and release characteristics of a spectrum of accidents. NUREG-0396, page I-37, states, "As can be seen from figure I-11, core melt accidents can be severe, but the probability of large doses drops off substantially at about 10 miles from the reactor.

In conducting the review of the TR, the NRC staff followed the Commission's expectations for risk-informed decision-making in the SRM to SECY-98-144. SRM-SECY-98-144 defines risk-informed decision-making, in part as:

> A "risk-informed" approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

RG 1.174, Revision 3, provides guidance on risk-informed changes in plant design and operation. RG 1.174, Revision 3, outlines five principles for implementing risk-informed decision-making for changes to a licensee's design or operation. While the NRC staff's review of the TR was informed by the five principles in RG 1.174, Revision 3, the objective of the review, consistent with NUREG-0396, is to ensure that adequate spectrum of accidents was identified for the purpose of EPZ sizing. Seismic risk is expected to dominate the risk to the public from the NuScale SMR design because of the reduced risk from internal events, external floods and high winds based on the risk profile of the NuScale DC (see Table 19.1-80 in Reference 26). Based on the NuScale design's risk profile, which the NRC staff expects to be representative of the NuScale design variations and derivatives, the NRC staff's review focused on the inclusion of seismic risk in the spectrum of accidents. The NRC staff's approval of this TR, regarding the use of PRA and risk insights is limited to the sole purpose of sizing the EPZ. It should not be extrapolated to risk-informed applications and decisions that impact NuScale's plant design, operations, and/or other emergency planning requirements.

The NRC staff issued RAIs 9828 and 9830 based on its review of Revision 2 of the TR. In addition to revising the TR to Revision 3, NuScale submitted its responses to RAI 9828 and RAI 9830 and incorporated the responses in Revision 3 of the TR.

The NRC staff performed a regulatory audit as part of its review of the methodology in the TR (Reference 27). The purpose of this audit was to gain a better understanding of the methodology and to review an example calculation that implemented key steps in the methodology.

3.2 Evaluation of Applicability of the TR to NuScale Design (TR Section 2.5.1)

In Section 2.5.1 of Revision 3 of the TR, NuScale stated that the TR is applicable to a NuScale SMR design, including the standard plant design (Docket 52-048), and variations and derivatives thereof comprising all the characteristics listed in TR Section 2.5.1 and in Section 1.0 of this SE.

The NRC staff's review finds that the characteristics listed in TR Section 2.5.1 will identify the NuScale SMR design with the features and risk profile that were considered by the NRC staff in its review. Therefore, the NRC staff finds that limiting the applicability of the TR per Section 2.5.1 of Revision 3 of TR-0915-17772 is appropriate and that meeting all the characteristics listed in Section 2.5.1 of Revision 3 of TR-0915-17772 is acceptable to demonstrate applicability of this TR to the NuScale SMR design, including the standard plant design (Docket 52-048), and variations and derivatives thereof.

3.3 Evaluation of Dose Criteria and Dose Evaluation Methodology

3.3.1 Accident Dose Related Criteria (TR Section 3.2)

The NRC staff compared the TR proposed dose-based criteria to the applicable technical basis for the establishing the EPZ size in existing regulations discussed above. Based on its review, the NRC staff finds the TR methodology to be acceptable and generally consistent with NUREG-0396.

The TR proposes accident dose criteria for EPZ sizing based on the NUREG-0396 sizing rationale, which is used as input to determine the generic distance for the plume exposure EPZ for existing plants. The TR approach considers design-basis accidents (DBAs), less severe beyond design basis accidents, and less probable but more severe beyond design basis accidents. Less severe accidents are those where the containment is expected to be intact. More severe accidents are those where the containment is expected to either fail or be bypassed. Each of these three accident classes are assessed against three dose-based figures-of-merit criteria. The TR dose-based criteria for EPZ sizing are as follows:

> Criterion a: The EPZ should encompass those areas in which projected dose from DBAs could exceed the early phase PAGs.
>
> Criterion b: The EPZ should encompass those areas in which consequences of less severe accident sequences could exceed the early phase PAGs.
>
> Criterion c: The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe accident sequences.

The methodology determines the final EPZ distance as the largest distance among the following:

- For Criterion a, the distance at which dose does not exceed either a 1 rem TEDE criterion at mean weather conditions or a 5 rem TEDE criterion at 95th percentile weather conditions for design-basis source terms (DBST),

- For Criterion b, the distance at which dose does not exceed either a 1 rem TEDE criterion at mean weather conditions or a 5 rem TEDE criterion at 95th percentile weather conditions for screened-in less severe accident sequences,

- For Criterion c, the distance at which the {{
  }} sequences, and

- The site boundary (i.e., the minimum EPZ).

Biological health effects of radiation are typically divided into two categories. The first category consists of exposure to low doses of radiation over an extended period of time producing chronic or long-term effects. The second category consists of exposure to high doses of radiation over short periods of time producing acute short-term effects. Low doses spread out over long periods of time do not cause an immediate problem to tissues and organs. The health effects of low doses of radiation occur at the cellular level, tending to damage or change them, and may not be observed for many years. High doses at high dose rates tend to kill cells and damage tissue and organ functionality. This, in turn, may cause a rapid whole body response often called the acute radiation syndrome. A class of acute radiation syndrome, hematopoietic syndrome, occurs in the red bone marrow and is clinically observed at the lower range of high dose and high dose rate exposures.

Criterion a and Criterion b of 1 rem and 5 rem TEDE over 96 hours (four days) are applied to DBAs and less severe accidents beyond the design-basis. For DBAs and less severe accidents, the 1 rem TEDE over 96 hours is the primary criterion and the 5 rem TEDE is used to confirm the analysis under unfavorable atmospheric transport and dispersion conditions. The NRC staff finds this acceptable since the numerical values of 1 rem and 5 rem TEDE are consistent with the EPA PAG Manual early phase PAGs, which establish dose criteria as a range from 1 to 5 rem TED over four days. The 1 rem TEDE utilizes mean meteorology and 5 rem TEDE utilizes 95th percent meteorology. The 1 rem TEDE is the primary criterion as it represents a lower dose criterion, which conforms with the EPA PAG Manual recommendations for best-estimate modeling. The TEDE evaluates the internal deposition of radioactivity, which includes the prospective dose component arising from retention of radionuclides in the body beyond the period of environmental exposure, i.e., the dose commitment. The dose commitment is evaluated over a period of 50 years to protect against chronic or long-term effects.

For Criterion c, the TR uses the metric of 200 rem whole body acute dose over 24 hours based on a substantial reduction in early severe health effects such as acute radiation syndrome or hemopoietic syndrome, which is consistent with NUREG-0396. This criterion is applied to BDBAs that involve containment bypass, termed “more” severe accidents in the TR. Criterion c is considered met when {{

}}. The NRC staff notes that WASH-1400, and subsequently NUREG-0396, calculated dose in terms of whole body dose equivalent, which is not a contemporary practice for radiological consequence analyses to estimate acute health effects. Reference 28 recommends the use of the effective dose equivalent versus the whole body dose to evaluate compliance with radiation protection standards in the workplace. However, the effective dose equivalent is not appropriate for high exposures that may be encountered with reactor accidents. Where the whole body dose equivalent had been used in previous evaluations, the red bone marrow dose equivalent has now replaced the whole body dose equivalent (References 29 and 30). The value of 200 rem acute dose to the red bone marrow is generally consistent with the lower range of clinically observable acute radiation health effects (see Table 2.1 of Reference 31).

The TR states that Criterion c is met if {{

}}. The TR states that this is consistent with the level of protection provided to the public by the 10-mile EPZ determined in NUREG-0396. The NRC staff’s review notes that NUREG-0396 determined that the 10-mile plume exposure EPZ would be appropriate based in part upon the information presented in Figure I-11 of NUREG-0396. This figure presents a set of curves representing the conditional probability of exceeding whole body dose versus distance. These curves were produced based on release frequencies and source terms from WASH-1400 and were normalized based on an estimated core damage frequency of 5E-5 per reactor year. NUREG-0396 acknowledges that probability of exceeding EPA PAG doses at 10 miles is 1.5E-5 per reactor year (one chance in 50,000 per reactor-year). The NRC staff’s review notes that based on Figure I-11 in NUREG-0396, {{

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NUREG-0396 also identifies the shape of the 200-rem curve as being important to the decision on EPZ size. Based on Figure I-11, the authors of NUREG-0396 state that core melt accidents can be severe, but the probability of large doses drops off substantially at about 10 miles from the reactor.

Based on its review, the NRC staff determined the need for a “Condition of Use” to ensure that the EPZ size determination using the TR methodology {{

}} demonstrating that Criterion c is met. As stipulated in Section 5.0 of this SE (Condition G), the {{ }} used to satisfy Criterion c in Section 3.4.2 of this TR will be monotonically decreasing and demonstrate that the probability of large doses drops off substantially. Based on its review discussed above, the NRC staff finds the {{ }} for demonstrating that Criterion c is met to be acceptable because, in conjunction with the NRC staff’s corresponding “Condition of Use” discussed above, it is consistent with NUREG-0396.

In application, the DBAs utilizing the design-basis source term will be evaluated against Criterion a. The EPZ distance, as calculated by the design-basis source term, will be compared to the EPZ distances from the less, and more severe, accident sequences screened in from the PRA (see TR Section 3.4), which are compared against Criterion b and Criterion c. The final EPZ distance will be determined as the largest distance among those based on Criteria a, b, c, and the site boundary.

3.3.2 Source Term and Dose Evaluations (TR Section 4.0)

3.3.2.1 Application of Software (TR Section 4.1)

The NRC staff evaluated the proposed use of software applications to derive design-specific source terms to assess severe accidents and radiological consequence when evaluating plume exposure EPZ sizes. To perform such analyses, software applications would meet the following TR-defined criteria:

> Calculate time-dependent source terms to the environment that reflect the spectrum of accident sequences.
>
> Calculate site-specific offsite dose consequences for the time-dependent source terms to the environment.

The recommended software for source term and dose evaluations is described in TR Section 4.1. The methodology recommends the use of the RELAP5, MELCOR code (Methods for Estimation of Leakages and Consequences of Releases) coupled with MELMACCS (MELCOR to MACCS interface) to generate the source term for use in the MACCS (MELCOR Accident Consequence Code System) computed code. Section 4.1 of the TR states that use of any computer codes different from those recommended should be technically justified and cover the same range of phenomena.

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants. It is considered a second-generation plant risk assessment tool, which can model a broad spectrum of severe accident phenomena in both boiling and pressurized water reactors (PWRs). These include thermal hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat-up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications. RELAP5 is a tool for analyzing small-break loss-of-coolant accidents (LOCAs) and system transients in PWRs or boiling water reactors (BWRs) and can be used to confirm certain aspects of the MELCOR model. It has the capability to model thermal hydraulic phenomena in one-dimensional volumes. Additionally, TRACE is a modernized thermal-hydraulics code designed to consolidate and extend the capabilities of the NRC's three legacy safety codes - TRAC-P, TRAC-B, and RELAP. TRACE is able to analyze large/small-break LOCAs and system transients in both PWRs and BWRs. The capability exists to model thermal-hydraulic phenomena in both one-dimensional and three-dimensional space.

To perform site-specific offsite dose consequences, the TR requires the latest version of the NRC’s MACCS code. MACCS is a fully integrated, engineering-level computer code. MACCS simulates the impact of severe accidents at nuclear power plants and other nuclear facilities on the surrounding environment. MACCS can use site-specific weather data to determine hypothetical land contamination levels, doses to individuals, health effects and risks on populations based on protective action recommendations, and economic losses resulting from a nuclear power plant accident.

The NRC staff confirms that the latest version of MELCOR, RELAP5, MACCS, and various associated codes are appropriate tools for assessing design- and site-specific severe accidents and radiological consequences when evaluating plume exposure EPZ sizes. The NRC staff also confirms that the MELCOR and MACCS codes are capable of performing analyses, which produce results that can be compared to the figures-of-merit dose-based criteria discussed above. Additionally, the use of deposition velocities estimated by MELMACCS and short duration plume segments is recommended. The NRC staff considers the use of the MELMACCS code to process MELCOR information to be reasonable because MELMACCS has been specifically designed to serve as an interface between MELCOR and MACCS and will generate a radiological source term for use in MACCS that is consistent with the accident progression modeling. Specifically, MELMACCS will ensure that the plume segment information needed by MACCS contains information on the onset and duration of plume segments, the release fractions for each plume segment, and the location and buoyancy of each plume segment that is consistent with the accident progression modeling in MELCOR. MELMACCS core inventory files appropriate for the NuScale design may not be available as part of a typical MELMACCS distribution and may require independent development.

The NRC staff expects users of the TR to implement best-practice-modeling approaches when utilizing these codes and document the specific manner-in-which their models represent important and, in some cases, uncertain aspects of severe accident behavior and radiological consequences. The NRC staff expects this documentation to include choices made among alternate modeling options offered through each of the code inputs, changes to selected input parameters from those offered as 'default' values, and, in some cases, user-generated 'models' to represent features of plant response to a severe accident and offsite emergency response planning that are not directly available in the codes. For each of these codes, the NRC staff and its contractors have developed several helpful documents, which contain the applicable code model descriptions, verifications and validation reports, manuals, and user guides. The NRC staff expects that these documents to be considered by the user of the TR in evaluating plume exposure EPZ sizes in any submittal. The TR states that, upon request, a COL applicant should provide input and output files for each code used to the NRC to facilitate its review. The NRC staff agrees that, upon request, a COL applicant should provide input and output files for each code used to the NRC to facilitate its review.

3.3.2.2 Source Term and Dose Evaluation Methodology (TR Section 4.2)

The NRC staff evaluated the TR proposed methodology for performing evaluations of the source term and assessing radiological consequences when determining an appropriate EPZ size basis. When performing the analyses, applicants shall demonstrate the acceptability of the EPZ size by meeting the criteria discussed in Section 3.3.1, above. The TR, Table 4-1, provides a high-level summary of the source term and radiological consequence methodology for evaluating plume exposure EPZ sizes.

3.3.2.3 Source Term Methodology

The NRC staff reviewed the selection of release scenarios for which doses would be assessed when developing source terms. The "source term" denotes the radioactive material composition and magnitude, as well as the chemical and physical properties of the material within the containment that are available for leakage from the reactor to the environment. The TR methodology identifies a range of scenarios ranging from DBAs (as discussed in Section 3.3 of the TR) to BDBAs (as discussed in Section 3.4 of the TR). This is consistent with NUREG-0396, which considered a spectrum of accident conditions from DBAs and BDBAs.

The TR, Section 3.3, discusses the methodology for use of design-basis source terms developed for traditional Chapter 15 Transient and Accident Analysis of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (Reference 32) contained within the facility’s FSAR to represent DBAs. This is consistent with the approach described on page I-5 of NUREG-0396, which states:

> The design-basis loss-of-coolant accident (DBA-LOCA) has been typically the most severe design-basis accident in that it results in the largest calculated offsite doses of any accident in this class . . . An analysis of this accident has been performed for most of the power plants licensed or under review by NRC to determine the dose/distance relationships as computed by traditionally conservative assumptions used under 10 CFR Part 100 requirements.

In Section 3.4 of the TR, NuScale describes a process for identifying accident sequences from the PRA and then screening them for inclusion in EPZ sizing analysis for BDBAs. The compilation of sequences is described in Section 3.4.1. The screening of seismic single module accident sequences {{ }} is described in Section 3.4.2. The screening of non-seismic single module accident sequences based on the sequence core damage frequency is presented in Section 3.4.3. The screening process involves both the absolute magnitude of the sequence frequency, with non-seismic sequences greater than or equal to 1E-7 per module-year being retained for further analysis in the EPZ technical basis. {{ }} The NRC staff notes that the explicit application of screening criteria to determine whether to include or exclude specific sequences does not appear to be a feature of the NUREG-0396 analyses. NUREG-0396 does not appear to have excluded any of the WASH-1400 release categories. For example, on page I-37 of NUREG-0396 states,

> A set of such curves has been prepared for all of the RSS [Reactor Safety Study] accident release categories (figure I-11). These curves include both Pressurized and Boiling Water Reactor (PWR & BWR) accidents.

Details of the release categories from WASH-1400 can be found in Table VI 2-1 of WASH-1400 (Reference 33). Examination of that table shows that the lowest frequency release category was PWR-4, which is a containment isolation failure with a frequency of 5E-7 per reactor year. It should be noted that the process provided for screening non-seismic sequences in TR Section 3.4.3, with a screening level of 1E-7 per module year, would have screened in all the WASH-1400 sequences, since none of these non-seismic sequences had a frequency less than 1E-7 per reactor year. The NRC staff’s findings on the acceptability of the seismic and non-seismic screening thresholds for the purposes of EPZ sizing (i.e., this application) are provided in Section 3.5 of this SE.

To evaluate consistency with the NUREG-0396 approach for determining which dose levels were appropriate for evaluating different types of accident sequences, the NRC staff evaluated how the TR categorized accidents to determine which dose criteria apply for specific families of accident sequences. The categorization of the accident sequences into less severe and more severe is described in TR Section 3.4.5. These are differentiated based on containment integrity, with those sequences characterized by a loss of containment integrity considered "more severe" and those in which containment integrity is not impaired being considered less severe. If containment integrity for a given sequence is uncertain, consideration is to be given of both the less severe and more severe versions of the accident sequence. This classification of severity is based on the level of atmospheric release from containment, where more severe accidents involve a significant atmospheric release of radionuclides to the environment due to a compromised containment, and less severe accidents involve much smaller atmospheric releases due to an intact containment barrier. This concept is broadly consistent with the differentiation used in NUREG-0396. As described on page I-6 of NUREG-0396:

> The lower range of the spectrum would include accidents in which a core "melt-through" of the containment would occur. As in the DBA-LOCA class, the doses from "melt-through" releases (involving thousands of curies) generally would not exceed even the most restrictive PAG beyond about 10 miles from a power plant. The upper range of the core-melt accidents is categorized by those in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere because of over-pressurization or a steam explosion.

Section 3.4.5 of the TR states that the frequency of radionuclide release associated with each sequence should be used in the downstream dose evaluation. This is consistent with the use of the release frequencies for dose-distance calculations and with the approach in NUREG-0396.

The TR stipulates that when performing the accident source term evaluation, the user of the TR would develop an integrated severe accident model utilizing the design- and site-specific PRA accident sequences and screening criteria previously discussed. The purpose of this evaluation would be to determine fuel/cladding failure with their associated radionuclide release fractions and timing. This model would need to include, at a minimum, the primary coolant system, containments, important safety and non-safety-systems, and associated control logic. As EPZ evaluations may credit mitigative design features, the model would likely include unique design features that can mitigate the release of radioactive material to the environment. For design features that are not safety related, the NRC staff expects the user of this TR to provide detailed technical justifications. The TR also encourages its user to employ separate-effects models to increase the fidelity of the severe accident simulations, while simultaneously decreasing the computational burden of a larger integral model. The NRC staff agrees that separate effects models can increase fidelity of severe accident simulations.

3.3.2.4 Assessing Offsite Radiological Consequences

Assessing offsite radiological consequences is an integral part of determining an appropriate EPZ size. NuScale's approach as reflected in the TR is that demonstrating an appropriate EPZ size requires the applicant to utilize the MACCS code to estimate offsite dose consequence evaluations with certain modeling assumptions and meet specific criteria listed in Section 4.1.2 of the TR, and as provided below:

- Modeling a stationary population without credit for protective actions.
- Use of a year of meteorological data representative of the site analyzed, selected from a minimum of 3 years of available meteorological data.
- Use of the appropriate dose conversion factors (DCFs) for TEDE and acute red marrow dose in MACCS.

- Use of an atmospheric transport and dispersion (ATD) model accepted by the NRC for use in the presence of building wake effects (i.e., the Ramsdell and Fosmire building wake model in MACCS version 4.1 or later), or confirmation of atmospheric transport and dispersion results within 0.31 mi (0.5 km).

Modeling a Stationary Population

The NRC staff reviewed the TR guidance to determine how radiological exposures would be modeled for DBAs, less severe BDBAs, and more severe BDBAs. The dose evaluation methodologies for DBAs, less severe accidents (TR Section 4.2.2), and more severe accidents (TR Section 4.2.3) assume no protective actions are taken, and the NRC staff finds this approach conservative and consistent with NUREG-0396. The methodology also assumes that exposure durations of 96 hours will be applied to the dose assessments for DBAs and less severe accidents (Criterion a and Criterion b), and that an exposure duration of 24 hours will be applied to the dose criteria for more severe accidents (Criterion c). The TR methodology also includes dose projections with ad hoc protective measures in response to DBAs as an additional analysis that could be performed after initial EPZ size calculations have been conducted.

The methodology specifies that either the MACCS Type A peak dose output (peak dose at a distance) or the MACCS Type 6 peak dose output (peak centerline dose) is to be used. The methodology clarifies that both exposure to both airborne (from cloudshine and inhalation) and deposited (from groundshine and resuspension) radioactivity will be included. The TR recommended exposure assumptions are consistent with no protective actions for a period of either four days (for comparison to PAG levels), or 24 hours for reduction of early health effects. Key assumptions include:

- Use of a dose-in-place model with no evacuation,
- Exposure duration of 96 hours (345,600 seconds) for DBA and less severe BDBA,
- Exposure duration of 24 hours for more severe BDBA, and
- Shielding factors of 0.7 for groundshine (reflecting the roughness of actual terrain vs an idealized plane) and unity for cloudshine and inhalation).

The NRC staff finds that the use of a four-day exposure duration for DBA and less severe BDBA is consistent with the exposure duration recommended for the EPA Early Phase PAG levels discussed in the EPA PAG Manual and is longer than the exposure duration of 24 hours assumed in NUREG-0396. The use of a 1-day exposure duration for the more severe BDBA is consistent with the assumptions of NUREG-0396, as indicated by the caption to Figures I-15 and I-16 ("Whole body (thyroid) dose calculated includes: external dose to the whole body (thyroid) due to the passing cloud and 1-day exposure to radionuclides on ground, and the dose to the whole body (thyroid) from inhaled radionuclides within 1-year"). The NRC staff notes that the shielding factors are consistent with an individual located outdoors with no shielding apart from natural ground roughness, based on footnote a of Figures I-15 and I-16 ("*Shielding factor for airborne radionuclides = 1.0. Shielding factor for radionuclides deposited on ground = 0.7. 1-day exposure to radionuclides on ground*").

Meteorological Data

The NRC staff reviewed NuScale's methodology to determine its approach to using meteorological data in the analyses. Meteorological data considerations are described in the TR Section 4.1.2, where it is stated that:

> "A meteorological file should be created by obtaining meteorological data available for five years that is most representative of the meteorological conditions at the site, preferably at a location close to the site, and performing EPZ analyses separately for each year. The five years of data need not be consecutive. If five years of data are not available, a minimum of three years of data may be used. If three years of data are used, a statistical analysis of the distribution of stability classes for each of these three years should be performed to demonstrate that an adequate sample of site-expected meteorological data has been utilized."

While this appears to be reasonably consistent with the approach used in WASH-1400 that supported the NUREG-0396 severe accident analyses, the NRC staff expects applicants to refer to the appropriate RGs, which provide guidance concerning onsite meteorological measurement programs. The meteorological data used in WASH-1400, Appendix VI, states that "From the many reactor sites in the U.S., a total of seven broad types were selected as being representative of variability of climatic or topographic features." Although WASH-1400 was not a site-specific analysis, the use of different sites in the study to reflect local climatic or topographic features is consistent with the recommendation in the methodology to use data representative of the meteorological conditions of the site. The TR does not provide for an assessment of the quality and completeness of the meteorological data, but the methodology does indicate that missing data should be filled in using the approaches used in State-of-the-Art Reactor Consequence Analyses (SOARCA), which relied on the guidance documented in Reference 34. The NRC staff notes that for the assessment of releases from DBAs described in TR Section 4.2.1, the methodology requires use of MACCS with a weather file that samples every hour from the selected year of weather data. This is not strictly consistent with the use of conservative meteorological conditions for DBA analyses in NUREG-0396 inferred from the statements on pages I-26 – I-27 that:

> Inherent in the consequence calculations for the postulated DBA-LOCA is the presumption of "five percentile" [95th percentile] meteorology, i.e., the presumption that atmospheric dispersion at a site at the time of the postulated accident should be more favorable (leading to lower doses) ninety-five percent of the time…It can nominally be characterized by class F stability and very low wind speeds (e.g., 2 miles/hour or less), i.e., the very conditions for which a wind shift is most likely"

However, the use of dose results from the 95th percentile sampled meteorological conditions, for comparison to the upper 5 rem TEDE PAG level, is consistent with the intent of conservative meteorology.

Dose Conversion Factors

The TR recommends that the TEDE radiological acceptance criteria be computed using the most appropriate dose conversion factors (DCFs) included with the latest released version of MACCS.

The NRC's TEDE is based on ICRP dosimetry models of ICRP 26/30 where the tissue weighting factors for this system of dosimetry are directly codified in 10 CFR Part 20. Acceptable practices for computing radiological consequences in terms of TEDE are to apply the exposure-to-committed effective dose equivalent factors for inhalation of radioactive material found in Table 2.1 of Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*. The factors in the column entitled "effective" yield doses corresponding to the committed effective dose equivalent. These tables are derived from the data provided in ICRP Publication 30 and have been found acceptable to the NRC staff as they meet the applicable regulatory requirements. Likewise, the exposure-to-effective dose equivalent factors for external exposure of radioactive material apply Federal Guidance Report No. 12, *External Exposure to Radionuclides*. Therefore, by default, compliance with the dose-related regulations of Parts 50 and 20 is demonstrated when applying the exposure-to-DCFs of Federal Guidance Reports (FGRs) 11 and 12.

The TR cites Reference 35, which provides additional information identifying currently available MACCS DCF files that provide dose coefficients to compute TEDE. The MACCS "DOSDATA" and "FGRDCF" files may be used when dose coefficients based on Federal Guidance Report 11 and 12 are required and when estimation of acute health effects, or of latent health effects based on the full set of equivalent organ doses, is not needed. If used, the user should be aware that the DCF values in "FGRDCF" files also include contributions from short-lived decay progeny, and the nuclide list comprising the core inventory should be adjusted to eliminate decay progeny implicitly included in the dose coefficients when using these DCF files. The currently recommended DCF file to use, if dose coefficients are to be fully consistent with FGRs 11 and 12, would be Dosd60.inp file because it: a) contains the dose coefficients for the 60 radionuclides identified by NUREG/CR-4467 (Reference 36); b) contains an effective dose equivalent dose coefficient consistent with FGR-11 and FGR-12; and c) has chemical forms for inhalation that that are consistent with the recommendations of NUREG/CR-4185 and NUREG/CR-4691 (Reference 37). The NRC staff agrees with the TR that DCF files should either be unmodified, or any modifications be technically justified.

NuScale recommends in the TR that the early severe health effect metric of 200 rem whole body acute dose utilizes the MACCS "red bone marrow" output parameter (e.g., A-REDS MARR) as an acceptable surrogate for acute whole body dose. The NRC staff finds this acceptable as the use of the acute dose to the red bone marrow as a surrogate for whole body acute dose is consistent with the current usage in the MACCS code, which estimates acute doses only for specific organs and not for the whole body. Acute dose coefficients in MACCS are designed to consider the effect of dose protraction on acute health effects, whereas the chronic dose coefficients are used for estimating stochastic health effects from lifetime exposures. The NRC staff considers the use of an acute dose of 200 rem to the red bone marrow, as estimated by the MACCS code as an indicator of significant early health effects, to be consistent with the information on dose levels associated with acute radiation syndrome, or hemopoietic syndrome, provided in Table 3-2 of EPA PAG Manual.

<u>Atmospheric Transport Modeling</u>

The NRC staff reviewed the TR methodology to evaluate the recommended atmospheric transport model. Since the methodology requires the use of the MACCS code for dose calculations, the methodology implies the use of the current Gaussian plume segment atmospheric transport model used in MACCS. Application of the EPZ methodology using

MACCS would require parameter choices related to dispersion curves, treatment of near field effects, such as plume rise, plume meander, and building wake effects, and parameters to model dry and wet deposition. The source of parameter values for the atmospheric transport model used in MACCS is not clearly specified in the document, although TR Section 4.2.1 states,

> The EPZ boundary dose calculation will apply a methodology similar to that used in the SOARCA study (Reference 6.2.14 [of the TR]), which used MACCS state-of-the-art consequence analysis software. MACCS input parameters for the applicable site and design-specific source terms will be developed. An example of a site-specific MACCS model can be found in the Surry SOARCA study (Reference 6.4.5 [of the TR]).

Based on these statements, it appears that the methodology will rely on the approaches and parameters used in the SOARCA analyses, unless otherwise specified. The NRC staff considers this to be reasonable, but notes that more recent contractor reports have been published since SOARCA (e.g., Reference 38) contains modeling practices and parameter choices considered appropriate for the purposes of the SOARCA project; some of which may, or may not, be appropriate for the determination of an EPZ size from an SMR. Therefore, the NRC staff encourages the user of this TR to justify selected modeling practices and parameter selections in its application.

TR Section 4.2.4 addresses the use of the MACCS model at close distances (less than 500 m). The methodology states that Atmospheric Relative Concentrations in Building Wakes (ARCON)-96 will be used as a code, which NRC has accepted for use in evaluating control room habitability for dispersion conditions in the presence of building wakes. Additionally, the TR identifies, as an acceptable option, another recently NRC staff-approved building wake model by Ramsdell and Fosmire, which is packaged with MACCS version 4.1, or later. The staff finds acceptable the TR recommendations for atmospheric transport modeling packaged with the MACCS code. The methodology states:

> The ratio of the MACCS X/Q to the NRC accepted code X/Q values for mean and 95th percentile X/Q, respectively, shall be compared at each distance up to 500 meters. These mean and 95th percentile X/Q ratios shall be greater than or equal to 1.0 at the minimum distance to the site boundary and 500 meters for the MACCS results to be considered acceptable within 500 meters.

This methodology states that the comparison will be carried out by modeling a ground level release of a plume with no thermal energy, but that treatment of building wake effects “should follow the respective modeling best practices for both codes.” The NRC staff notes that it has been customary for reactor severe accident consequence analyses to use an approach that employs the virtual source approach described on pages 33-34 of Reference 39 in which an initial plume dimension is assigned based on the characteristics of the building from which the release occurs. The staff considers this approach reasonable because the ARCON-96 model is based on empirical data collected at close distances, subject to the effect of building wakes, and because the comparison will be used to ensure that the MACCS results are conservative with respect to the ARCON-96 results. The TR methodology appropriately notes that MACCS parameters related to building wakes and plume meander will be the factors to be adjusted to ensure that the MACCS results are bounded by the ARCON-96 results.

3.3.2.5 Uncertainty Analysis Methodology for Source Term and Dose Calculations

The NRC staff reviewed the TR methodology to evaluate uncertainty using the TR EPZ sizing methodology described in the TR Section 4.3. The TR framework for EPZ sizing uncertainty analysis is based on the NRC methods published in the report entitled, "SOARCA Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station" (Reference 40). The purpose of the uncertainty analysis is to understand the important sources of uncertainty in the technical basis for EPZ size and to provide additional confidence in the best-estimate parameters used. Application of the uncertainty analysis methodology is provided in the TR Section 4.3, Figure 4-4, with a series of steps. These steps cover the: selection of accident sequences; identification of source terms and dose calculation uncertain parameters and their distributions; utilization of Monte Carlo methods to sample uncertain parameters within the MELCOR and MACCS analyses to identify important parameters that contribute more than 5 percent of the total regression metric; and confirmation of best-estimate values for important parameters. The NRC staff finds this uncertainty methodology to be a reasonable approach to understanding the important sources of uncertainty, when deriving site-specific EPZ for the NuScale design.

3.4 Evaluation of Defense-in-Depth (TR Section 3.7)

TR Section 3.7 discusses the defense-in-depth evaluation in the methodology. This section states that the methodology includes a qualitative evaluation of the seven defense-in-depth considerations in RG 1.174 as well as the five levels of defense identified by the International Atomic Energy Agency (IAEA) in INSAG-10, "Defense in Depth in Nuclear Safety" (Reference 41). This evaluation is independent and separate from the accident screening process and is performed to confirm that the design and operation maintain consistency with the defense-in-depth philosophy.

Consistent with the SRM to SECY 98-144 and RG 1.174, Revision 3, the NRC staff's review included evaluation of defense-in-depth. SRM to SECY 98-144 states that defense-in-depth is an element of the NRC's safety philosophy, which ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. RG 1.174, Revision 3, provides seven defense-in-depth considerations for risk-informed licensing actions.

The NRC staff's review determined that the defense-in-depth assessment discussed in Section 3.7 of the TR will result in the identification of key plant design and operational characteristics necessary for achieving defense-in-depth, including consideration of severe accident management strategies and diverse and flexible coping strategies. Based on its review, the NRC staff finds the treatment of defense-in-depth in the TR methodology to be acceptable because it is consistent with the intent of the Commission's PRA Policy Statement and the consideration of defense-in-depth in risk-informed decision making as discussed in RG 1.174, Revision 3. It should be noted that the NRC staff's finding does not imply a generic endorsement of INSAG-10; rather, it represents that, in combination with the defense-in-depth considerations in RG 1.174, Revision 3, the information in INSAG-10, supports a defense-in-depth evaluation for this application.

3.5 Evaluation of Safety Margins

The principles of risk-informed decision making in RG 1.174, Revision 3, includes consideration of sufficient safety margin. RG 1.174, Revision 3, states that with sufficient safety margins: (1)

the codes and standards or their alternatives approved for use by the NRC are met, and (2) safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met or proposed revisions provide sufficient margin to account for uncertainty in the analysis and data.

The TR does not include an explicit discussion of safety margins. The NRC staff's review determined that the methodology in the TR does not impact any codes or standards approved for use in the NuScale design. The methodology does not change the safety analysis acceptance criteria for the NuScale design. Therefore, the NuScale design's ability to meet or continue to meet approved codes or standards and its safety analysis acceptance criteria remains unchanged by this TR. Therefore, the NRC staff finds that safety margins in the NuScale design are not impacted by the methodology in this TR and an explicit discussion on this topic in the TR is unnecessary.

3.6 Evaluation of the Use of PRA Information and Insights with Consideration of Uncertainty (TR Sections 2.5.2, 3.4, and 3.8)

To evaluate a spectrum of accidents consistent with NUREG-0396, the TR states that PRAs used for the TR methodology will contain all hazards and all modes. The NRC staff's review focused on: (1) the use of PRA information and insights in the methodology consistent with NUREG-0396, the Commission's PRA Policy Statement, and SRM to SECY 98-144, (2) the guidance and practice on technical acceptability of the PRAs used for this risk-informed application, and (3) the consideration of PRA uncertainty consistent with the Commission's 1995 PRA Policy Statement, RG 1.200, and NUREG-1855, Revision 1.

The NRC Final Rule on Emergency Planning dated August 19, 1980 (Reference 42), pre-dated the Commission's Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement" (Reference 43). Consistent with NUREG-0396 and the Emergency Planning Final Rule of 1980, a comparison to the Commission Goals is not required for the purposes of EPZ sizing. However, the results and insights of a PRA submitted with a COL application by regulation or an operating license application under 10 CFR Part 50, consistent with Commission Guidance in the SRM to SECY 15-002, will be evaluated by the NRC staff to ensure that the PRA demonstrates that the Commission Goals for new reactors are met as directed in the SRM to SECY 90-016, ensuring the Commission Safety Goals for the design are met. The NRC staff will also ensure that the PRA results and insights submitted by a COL applicant, or an operating license applicant is consistent with the Commission's Severe Accident Policy Statement of 1985 for the design. Therefore, the NRC staff's review of the use of PRA information and insights for this TR is limited and applicable only to the risk-informed methodology in the TR for EPZ sizing. It is not applicable to, and does not extend to, either the review of the NuScale design or to the risk-informed applications related to the design and operation of the NuScale reactor.

<u>Technical Acceptability of the PRAs Used to Identify the Spectrum of Events in the Methodology</u>

Section 2.4, "NuScale Approach," of the TR states that the PRAs used in the application of the methodology in the TR must be of appropriate scope and technically acceptable. It further states that RG 1.200 describes one approach for determining the acceptability of a PRA. Section 2.5.2, "PRA Conditions," of the TR lists two conditions that are related to PRA technical acceptability: the PRA addresses internal and external hazards and all operating modes; and the PRA is demonstrated to be technically acceptable for the purpose of the TR methodology. A footnote to the second condition explains that certain items that demonstrate technical acceptability per the

guidance in RG 1.200, such as operational experience and plant walk downs, would not be feasible for a new nuclear power plant design.

Section 3.4.1, "Compilation of Probabilistic Risk Assessment Accident Sequences," of the TR, states that PRA accident sequences for all internal and external events, covering all operating modes, are compiled. In the TR methodology, a sequence refers to the progression from an initiating event to an end state within an event tree, each sequence representing a unique accident progression. The methodology treats all sequences individually within the accident selection process. Section 3.4.1 of the TR explains that any combination of failures of individual components or subcomponents within a system that fails the system mission in a top event in the PRA is considered to belong to the same sequence. The methodology allows for the user to group the screened-in sequences into release categories to reduce the number of source term and dose consequence analyses, provided the grouping is identified and justified. Section 3.4.1 of the TR states that any mitigation beyond the EOPs [emergency operating procedures], such as the SAMGs [severe accident management guidelines] and EDMGs [extensive damage mitigation guidelines], should not be credited in the frequency screening because the probability of these human actions has been historically difficult to quantify.

TR Section 3.5, "Other Risks" discusses the identification and evaluation of those non-PRA risks that may lead to potential off-site radionuclide release to ensure an appropriate planning distance. Examples of such "other risks" identified in Section 3.5 of the TR are volcanic activity and dropping the upper portions of the NuScale reactor pressure vessel and the containment vessel as they are moved to or from the dry dock area onto the fuel in the lower Reactor Pressure Vessel (RPV), which remains in the refueling flange tool, potentially causing mechanical fuel damage with the potential for a gap release. The TR states that applicants using the methodology need to demonstrate acceptable consideration of such risk contributors by meeting the dose-based criterion in Section 3.2 of the TR or by demonstrating that consequences of each identified "other risk" are bounded by screened-in events.

The NRC staff reviewed the above-described information using the guidance in RG 1.200, Revision 3, and DC/COL-ISG-028. The NRC staff's review of the TR determined that it includes a broad discussion about the need for technically acceptable PRAs to apply the methodology. Similarly, the "PRA Conditions" in Section 2.5.2 of the TR provide the use of the TR appropriate broad scope direction for PRA technical acceptability. The NRC staff also finds that the compilation of sequences from a PRA, including their grouping with identification and justification, and consideration of non-PRA risk contributors is appropriate for this methodology. In addition, the NRC staff's review determined that the TR methodology appropriately considers "other risks" in Section 3.5 of the TR.

Given the broad discussion on PRA technical acceptability in the TR, based on its review, the NRC staff determined that a "Condition of Use" on this TR is necessary to ensure that the technical acceptability of PRAs used in the TR methodology is consistent with the corresponding NRC staff guidance. Therefore, as stipulated in Section 5.0 of this SE, the "Condition of Use" for PRA technical acceptability for this TR is as follows:

1. The PRAs used for this TR will be design- and site-specific, and will be developed for all modes and hazards, including seismic events, using the guidance in RG 1.200, Revision 3, at Capability Category II. Any exceptions to meeting Capability Category II (e.g., inability to perform walkdowns) must be identified and justified by the user of the TR in its application.

2. The PRAs used for this TR will be peer reviewed in accordance with the guidance in RG 1.200, Revision 3.

3. The PRAs used for this TR will address hazards and modes not covered by endorsed consensus PRA standards, and the user will justify the technical acceptability of these PRA for use in identifying an acceptable spectrum of events.

The NRC staff expects the COL applicant using this TR to follow the endorsed PRA Standard and justify exceptions to meeting Capability Category II. The NRC staff's review of the PRA used for this TR will be informed by DC/COL ISG-028 to review the technical acceptability of the PRA in context of the EPZ application, except for the seismic PRA. ISG-028 does not cover seismic PRAs. The NRC staff will review seismic PRA used in this methodology, consistent with the above-mentioned "Condition of Use."

In summary, based on its review and in conjunction with the "Condition of Use" on PRA technical acceptability, the NRC staff finds that the PRAs used for the TR methodology will follow NRC staff guidance on PRA technical acceptability for risk-informed applications.

Evaluation of the Numerical Screening Threshold for all Hazards and all Modes Except for the Seismic Hazard

The user of this TR will develop an all hazards, all modes, design- and site-specific PRA with consideration of multi-module core damage sequences. As described in TR, Section 3.4.3, "Screening of Nonseismic Single Module Accident Sequences on Core Damage Frequency", the user of this TR would perform accident sequence screening based on core damage frequency for all hazards except for the seismic hazards. All non-seismic core damage accident sequences (i.e., sequences from internal hazards and non-seismic external hazards) with a point estimate core damage sequence frequency equal to or greater than 1E-7 per module year would screen in for the consequence analysis. The TR approach to screen out core damage sequences less than 1E-7 per module year is consistent with NUREG-0396. Specifically, the TR approach would have included all the WASH-1400 sequences based on Table VI 2-1 of WASH-1400. Therefore, the NRC staff finds the 1E-7 per module year core damage sequence screening threshold for non-seismic initiating events to be acceptable.

Section 3.4.3.1, "Parameter Uncertainty in Non-seismic Sequence Screening," describes how the methodology evaluates parametric uncertainty in non-seismic sequence screening. Figure 3-4 in the TR illustrates the non-seismic event screening process used in the methodology. The NRC staff's review of the parametric uncertainty evaluation for non-seismic sequences is provided in a subsequent sub-section of this SE.

As described in the TR Section 3.4.4, "Multi-Module Accident Methodology," applicants will demonstrate PRA acceptability by retaining multi-module accident sequences that meet the following criteria, as shown in Figure 3-5 of the TR:

- The sequence contains a site-wide or multi-module initiator, or a single-module initiator that physically propagates to other modules.

- Safe shutdown mechanisms are directly compromised in multiple modules, or a coupled safe shutdown function is failed.

- The resultant non-seismic multi-module accident sequences have a core damage frequency greater than or equal to 1E-7 per module year, including consideration of parametric uncertainty.

To prevent parsing of sequences into individual components for comparison against the numerical screening threshold, Section 3.4.1, “Compilation of Probabilistic Risk Assessment Accident Sequences” of the TR states that sequences used in the screening process are expected to be defined by an initiating event and top events representing the success or failure of mitigating systems at the system level. Any combination of failures of individual components or subcomponents within a system that fail the system mission in a top event are considered to belong to the same sequence. This ensures consistent application of the methodology. The NRC staff finds the multi-module sequence evaluation for non-seismic events acceptable because it appropriately identifies such sequences for source term and dose analysis in the methodology. The NRC staff finds that 1E-7 per module year core damage sequence screening threshold for non-seismic multi-module sequences to be acceptable since the 1E-7 per module year threshold is consistent with NUREG-0396, and the sequences will not be parsed into individual components for comparison against this threshold.

The NRC staff’s acceptance of the 1E-7 per module year sequence CDF screening threshold for this TR neither defines nor results in a regulatory finding on “credible” non-seismic events for the purposes of EPZ sizing or any other risk-informed decisions.

Evaluation of the Numerical Screening Threshold for Seismic Hazard

Section 3.4.2 of the TR explains that WASH-1400, which was used as the basis for NUREG-0396, identified external events risks, including seismic risk, and dispositioned these risks as comparable to risk from internal events. Therefore, seismic risk was not quantitatively evaluated either in WASH-1400 or NUREG-0396. {{

}}

As discussed in Section 3.4.2, “Screening of Seismic Single Module Accident Sequences {{ }} of the TR, the methodology will use {{ }} as the screening threshold for seismic events for inclusion in the EPZ sizing methodology in the TR. This means that seismic sequences that will be used for EPZ sizing purposes in the methodology will have {{

}}. In addition, Section 3.4.2 includes a {{

}} Figure 3-3 in the TR illustrates the seismic event screening process used in the methodology. The staff’s review of this {{ }} is provided in a subsequent sub-section of this SE.

Section 3.4.2 of the TR explains that seismic risk is assessed separately from non-seismic risks in the methodology because of the large uncertainties in the seismic hazard compared to internal events which are manifested in the seismic sequences due to the convolution of the seismic hazard with the event tree progression. Section 3.4.2 further states that a technically acceptable PRA, as necessitated by Section 2.5 of the TR, will be used.

The NRC staff's review determined that the {{

}} The NRC staff also performed an evaluation of the {{ }} to determine whether this threshold resulted in the inclusion of sufficient seismic BDBEs sequences for the NuScale design. The staff's evaluation used the following fundamental considerations:

- The objective of the EPZ, as explained in NUREG-0396, is to provide dose savings for a spectrum of accidents that could produce offsite doses in excess of the PAGs since no specific accident could be identified as the one for which to plan. This spectrum of accidents would need to include seismic events due to NuScale's risk profile discussed above.

- Consistent with NUREG-0396, the EPZ should be derived from the characteristics of design basis and beyond design basis accident consequences (termed "Class 9" events).
    - Class 9 accidents were defined in NUREG-0396 as those "considered to be so low in probability as not to require specific additional provisions in the design of a reactor facility," including total core melt scenarios "in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere."
    - As explained in NUREG-0396, it was understood that while these kinds of extreme accidents were unlikely, EPZs should be in place to provide defense-in-depth because "the probability of an accident involving a significant release of radioactive material, although small, is not zero."

- NUREG-0396 concluded that nuclear accidents were unique in important ways. The report explained that "the potential consequences of improbable but nevertheless severe power reactor accidents, while comparable in some sense to severe natural or man-made disasters, which would trigger an ultimate protective measure such as evacuation, do require some specialized planning considerations."

- Consistent with NUREG-0396, while the EPZ should not be solely dependent on the most severe and most improbable BDBEs, the determination of the EPZ size needs to include some of the key characteristics of very large releases.

- Consistent with NUREG 0396, the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects (injuries or deaths) in the event of the more severe Class 9 accidents."

The NRC staff's evaluation determined the seismic risk that would be expected to be excluded from EPZ sizing considerations (the "risk gap") based on the {{

}} The NRC staff evaluated the size of the "risk gap" based on nine plant sites

within the NuScale Certified Seismic Design Response Spectrum (CSDRS). Specifically, the NRC staff evaluated the percentage and absolute value of seismic CDF that would be excluded from EPZ sizing. Consistent with NUREG-0396, the objective of the NRC staff's evaluation was to ensure that a seismic risk was captured for consequence analysis to: (1) include an adequate amount of seismic events in the spectrum of accidents for EPZ sizing, and (2) to include some of the key characteristics of large releases from seismic severe accidents. Further, as discussed above, WASH-1400 and by extension NUREG-0396 did not include quantitative seismic risk estimates in the dose-distance calculations. Due to these constraints, the seismic CDF was used in the NRC staff's evaluation of the adequacy of the {{
}}

The NRC staff's evaluation and determination included the following steps:

Step 1: The NRC staff identified nine (9) operating reactor sites to provide a sufficiently large representation of seismic hazards across the contiguous US and collected the mean seismic hazard curves for these sites. The sites include the Western US (WUS) and the Central Eastern US (CEUS) with different site characteristics. The seismic hazard information used for each site includes the mean hazard curve at various spectral frequencies (e.g., 0.5 Hz, 1 Hz, 2.5 Hz, 5 Hz, 10 Hz, and 100 Hz) and the ground motion response spectrum (GMRS). The latest re-evaluated seismic hazard information was used for each selected site. The sites were selected such that the GMRS for each site was bounded by NuScale's CSDRS for the horizontal direction provided in Table 3.7.1-1 of the NuScale Final Safety Analysis Report (ML20224A484). Figure 1 shows the GMRS for the sites selected for this case study and the CSDRS. The GMRS was used to determine the spectral ratios representing the ratio of the acceleration at a particular frequency to the acceleration at peak ground acceleration (PGA; 100 Hz) (see ML100270582 for details on deriving spectral ratios). An example of the derived spectral ratios is provided in Table 2.

Table 1: Representative sites selected for NRC staff's evaluation

Site	Location	Site Characteristics
Plant 1'	Western US (WUS)	Soil
Plant 2'	Central Eastern US (CEUS)	Rock
Plant 3'	CEUS	Soil Backfill
Plant 4'	CEUS	Rock
Plant 5'	CEUS	Soil
Plant 6'	CEUS	Rock
Plant 7'	CEUS	Rock
Plant 8'	CEUS	Soil
Plant 9'	CEUS	Soil

Step 2: The NRC staff used the NuScale high confidence of low probability of failure (HCLPF) plant level fragility (PLF) of 0.84g anchored to the PGA. This HCLPF PLF is based on the NuScale FSAR Section 19.1.5.1.2.

Step 3: The NRC staff convolved the hazard curves at different frequencies with the PLF to estimate the seismic CDF. The convolution of hazard curve and PLF is a well understood and widely used method of estimating the seismic CDF (e.g., see Results of Safety/Risk Assessment of Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" ADAMS Accession number

ML100270582 and “Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates” ADAMS Accession number ML14083A586). The spectral ratio at each frequency determined in Step 1 was used to scale the PLF from PGA to the corresponding frequency (see ML100270582 for details of performing this scaling).

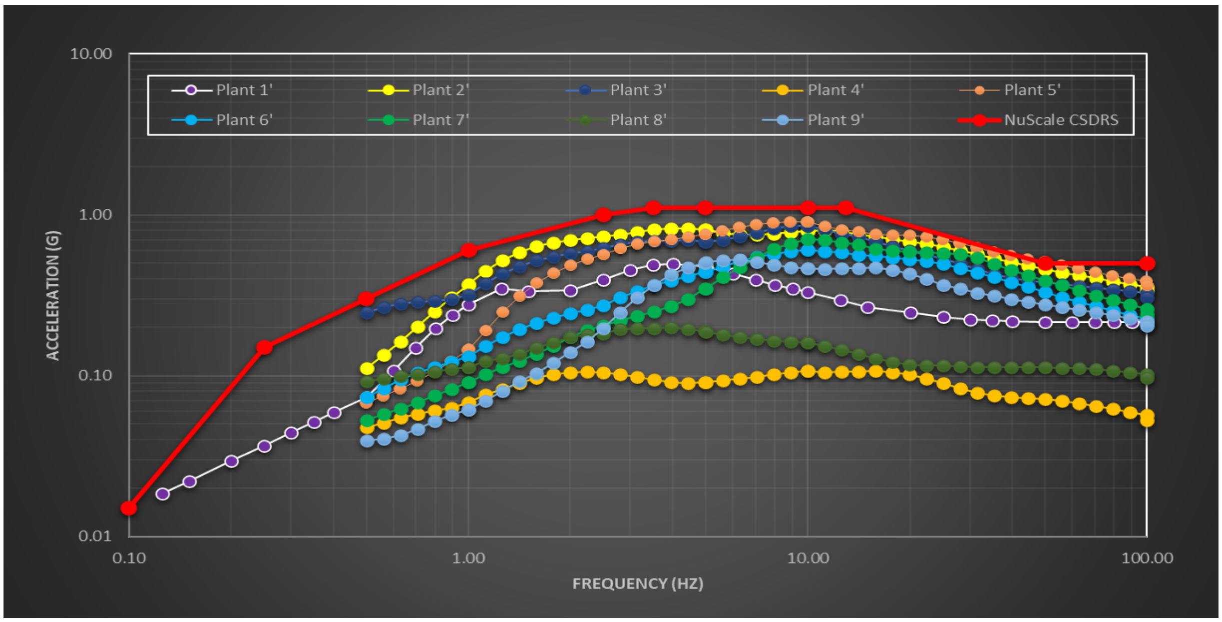

Figure 1: GMRS for the sites selected for the NRC staff’s evaluation and NuScale’s CSDRS

Table 2: Example of the derivation and use of spectral ratios:

Spectral Frequency (Hz)	Spectral Acc (g)	Spectral Ratio (-)	HCLPF at Spectral Frequency (g)	A_m at Spectral Frequency (g)
0.50	0.11	0.33	0.29	0.87
1.00	0.37	1.09	0.96	2.88
2.50	0.73	2.16	1.90	5.71
5.00	0.81	2.39	2.11	6.32
10.00	0.79	2.35	2.07	6.20
100.00	0.34	1.00	0.88	2.64

Step 4: The NRC staff calculated the simple average absolute “risk gap” below {{ }} for each site selected in Step 1. The absolute “risk gap” at each spectral frequency used in Step 1 is the seismic CDF below {{ }} A simple average of each of these

absolute “gaps” is used to determine the average absolute “gap.” The simple average implies an equal weight for each frequency. This is appropriate because the PLF is a representation of several SSCs and different SSCs are most susceptible to seismic events at different spectral frequencies. This is illustrated in Table 3. The total average seismic CDF is also calculated and used to determine the average relative “risk gap” (i.e., relative contribution, as a percentage, of the average “risk gap” to the average seismic CDF).

Table 3: Illustration of calculation of the “risk gap” at a particular spectral frequency

Spectral Acc (g)	Mean Exceedance Freq (per/year)	Conditional Failure Probability (-)	Seismic Core Damage Freq (per/year)
…	…	…	…
0.05	2.42E-03	2.03E-16	7.75E-20
…	…	…	…
0.10	8.14E-04	1.06E-11	1.47E-15
…	…	…	…
0.25	1.47E-04	8.78E-07	2.77E-11
0.35	6.90E-05	2.45E-05	4.01E-10
0.40	5.26E-05	6.66E-05	8.79E-10
0.50	2.92E-05	4.18E-04	3.31E-09
…	…	…	…
0.90	4.95E-06	1.61E-02	2.72E-08
1.00	3.26E-06	2.86E-02	3.30E-08
…	…	…	…
1.41	8.17E-07	1.18E-01	3.92E-08
1.77	2.80E-07	2.41E-01	3.00E-08
…	…	…	…
2.00	1.56E-07	3.22E-01	2.34E-08
2.24	8.27E-08	4.12E-01	1.70E-08
…	…	…	…

absolute
“risk gap”

The results from implementing the preceding steps are shown in Table 4. The calculated average absolute “risk gap” estimate is rounded to the nearest whole number in Table 4. The considerations used by the NRC staff to determine whether the evaluation results shown in Table 4 supported acceptance of the {{ }} were: (1) release frequencies in WASH-1400 and, by extension, NUREG-0396, (2) uncertainty in convolution-based estimates of seismic CDF, and (3) multi-module impacts.

Table 4: Absolute and relative “risk gap” based on the NRC staff’s evaluation for the selected sites

Site	Average Absolute “Risk Gap” Estimate (per year)	Average Relative “Risk Gap” Estimate (%)
Plant 1’	1.0E-07	29
Plant 2’	2.0E-07	53
Plant 3’	1.0E-07	68
Plant 4’	2.0E-09	100
Plant 5’	3.0E-07	47

Plant 6’	1.0E-07	82
Plant 7’	2.0E-07	76
Plant 8’	6.0E-09	99
Plant 9’	1.0E-07	81

As noted in Table 4, the relative “risk gap” is on average between 50% and 60%, which means that 40% to 50% of the seismic risk at these sites is included in the EPZ sizing decision. This is consistent with the foundational principles discussed earlier. Further, the maximum absolute “risk gap” in release frequency is approximately 3.5E-7 per year. This is comparable to the lowest frequency for the release fractions used in WASH-1400 and consequently, NUREG-0396, which was 4E-7 per year. Therefore, the release frequencies considered in EPZ sizing using the {{ }} are consistent with those considered in WASH-1400 and included in the NUREG-0396 evaluation. At some selected sites, the relative “risk gap” is about 80%, which means that 20% of the seismic risk at these sites is included in EPZ sizing decisions. However, at these sites, the absolute “risk gap” in release frequency is approximately 1E-7 per year. Therefore, the NRC staff’s evaluation determined that for these sites increasing the amount of seismic risk, which is included in the EPZ sizing decisions, e.g., from 20% to 50%, would not significantly impact EPZ sizing. Table 4 also shows that at low seismicity sites, the entire seismic risk is excluded from consideration at {{ }} The NRC staff’s evaluation determined that the total seismic risk for these sites is very low, as illustrated by the absolute “risk gap” for these sites in Table 4. Consequently, excluding the entirety of the seismic risk will not negatively impact EPZ sizing for these sites.

NuScale’s PLF HCLPF, which was also used for the NRC staff’s evaluation, is approximately 2.5 – 3.0 times higher than that for the majority of operating reactors. Since the HCLPF represents the seismic acceleration at which there is 95% confidence that the failure probability is 5% or less, the high HCLPF value for the NuScale design results in low absolute seismic risk at several sites. Consequently, attempting to capture larger amounts of seismic risk for consideration in EPZ sizing will be inconsistent with the principles in NUREG-0396 and result in an unnecessarily restrictive seismic event screening threshold for the NuScale design. The NRC staff’s review also determined that the {{ }} at the sites selected in the NRC staff’s evaluation is comparable to that used in the qualitative evaluation of seismic events in NUREG-0396.

Based on these reasons discussed above, the NRC staff’s evaluation supports the finding that the proposed {{ }} is acceptable because (1) it will result in the inclusion of seismic events, including beyond design basis seismic events, in the spectrum of accidents, and (2) it is consistent with the fundamental considerations for the staff’s evaluation, which are based on NUREG-0396.

Past comparisons of seismic CDF determined from seismic PRAs and the convolution method has demonstrated that while the convolution method provides close estimates of the seismic CDF from technically acceptable seismic PRAs, it can both over- and under-estimate the seismic CDF compared to technically acceptable seismic PRAs. Further, the differences in the estimates of seismic CDF from seismic PRAs and the convolution method have been found to be primarily due to differences in the expected and the realized plant-level fragility because the same seismic hazard is used in both cases. Since the NuScale design is expected to be developed with a target plant-level fragility in accordance with regulatory guidance, a large deviation between seismic CDF estimates from a technically acceptable seismic PRA and convolution method is not expected for NuScale. Consequently, while the uncertainty in convolution-based estimates is acknowledged qualitatively, the NRC staff did not adjust the estimates from its evaluation in either direction. The NRC staff notes that the methodology includes an evaluation to address uncertainty in {{ }}

As described in the TR Section 3.4.4, “Multi-Module Accident Methodology,” and illustrated in Figure 3-5 of the TR, the methodology identifies seismically initiated multi-module accident sequences using the following criteria:

- The sequence contains a site-wide or multi-module initiator, or a single-module initiator that physically propagates to other modules.

- Safe shutdown mechanisms are directly compromised in multiple modules, or a coupled safe shutdown function is failed.

However, in contrast to non-seismic events, the methodology excludes seismic sequences from the 1E-7 per year screening criteria for multi-module considerations. This is done to avoid additional screening of the seismic sequences because these sequences are initially screened based on {{ }}

The NRC staff’s evaluation also considered multi-module events qualitatively based on whether the seismic sequences included in EPZ sizing with {{ }} would have multi-module impacts. Seismic events are a common mode initiator and, therefore, by their nature, can result in multi-module impacts. Based on the results of the NRC staff’s evaluation, {{ }} Further, based on available information for the NuScale design in Section 19.1.5.1.2 of the NuScale FSAR, the dominant contributors for seismic risk are structural failures that are expected to impact multiple modules. Therefore, the NRC staff’s evaluation and NuScale’s seismic risk profile indicate that {{ }} will include events that have multi-module impacts. In addition, the source term for these sequences will also reflect multi-module impacts (e.g., by multiplying the source term with the number of modules impacted by a sequence). Therefore, the NRC staff finds that {{ }} in conjunction with the multi-module sequence evaluation in Section 3.4.4 of the TR is acceptable and will include seismic sequences with multi-module impacts.

Section 5, Stage C-1.2, of NUREG-1855 identifies {{ }} for use when screening is performed based on the {{ }} Further, {{ }} contains information about the uncertainty in {{ }} because of the lognormal distribution of {{ }} As a result, {{ }} is typically in the higher percentiles of {{ }} distribution. {{ }}

In summary, the NRC staff finds that {{ }} to be acceptable for this methodology and for the scope of applicability provided in Section 2.5 of the TR because: (1) it results in the inclusion of seismic events, including beyond design basis seismic events, in the spectrum of accidents for EPZ sizing for the NuScale design in the methodology, consistent with the approach and consideration of beyond design basis accidents in NUREG-0396, and (2) it results in the inclusion of multi-module sequences for the NuScale design.

The purpose, scope, and applicability of the NRC staff's evaluation and acceptance of {{ }} is limited solely to the methodology in this TR and the scope of applicability provided in Section 2.5 of the TR. It is not applicable and should not be used for screening seismic risk for purposes of seismic PRA development or for risk-informed decisions using seismic PRAs that impact reactor design and/or operations. The NRC staff's acceptance of {{ }} for this TR neither defines nor results in a regulatory finding on "credible" seismic events for the purposes of EPZ sizing or any other risk-informed decisions.

Based on its review and evaluation, the NRC staff determined the need for a "Condition of Use" that ensures the use of {{ }} for only those sites where the GMRS was bounded by the NuScale CSDRS. As stipulated in Section 5.0 of this SE (Condition D), the use of the {{ }} in this TR is allowed only if the site-specific probabilistic GMRS is bounded by NuScale's CSDRS for the horizontal direction provided in Table 3.7.1-1 of the NuScale FSAR (See Section 5.0 of this SE for citation. The GMRS used to demonstrate that this condition is met will be derived using the approach in RG 1.208, Revision 0 (See Section 5.0 of this SE for citation), and the site-specific probabilistic seismic hazard analysis accepted by the NRC as part of a COL or operating license application.

Based on its review and evaluation, the NRC staff determined the need for a "Condition of Use" that ensures the use of the {{ }} only if the HCLPF PLF is equal to or greater than 0.84g PGA. As stipulated in Section 5.0 of this SE (Condition E), the use of {{ }} in this TR is allowed only if the HCLPF PLF is equal to or greater than 0.84g at peak ground acceleration (100 Hertz). The HCLPF plant-level fragility will be the peak ground acceleration at the probability of 0.01 on the mean conditional core damage probability curve derived from a seismic PRA that meets the "Condition of Use" on PRA technical acceptability discussed above (Condition B in Section 5.0 of this SE).

Based on its review and evaluation, the NRC staff determined the need for a "Condition of Use" to ensure that Conditions D and E are met both at the time of submission of an application using the TR methodology and prior to fuel load to account for any changes in the design or construction between the submission of an application using the TR methodology and operation of the plant. As stipulated in Section 5.0 of this SE (Condition F), the user of this TR will demonstrate that Conditions of Use D and E, as described above in Section 5.0 of this SE, are met at the time of submission of an application using this methodology and prior to fuel load and the demonstration prior to fuel load will be based on the as-built plant.

<u>Treatment and Analysis of Uncertainties in PRA</u>

<u>Uncertainty in the Numerical Screening Threshold for All Hazards and All Modes Except Seismic Hazard</u>

As described in Section 3.4.3.1 of the TR, uncertainty in the screening of non-seismic core damage sequences against the numerical sequence screening threshold of 1E-7 per module year is informed by NUREG-1855, Revision 1. The mean value of the sequence frequency is compared against the numerical screening threshold of 1E-7 per module year. The sequence frequency is then categorized into regimes based on the proximity of the frequency to the 1E-7 per module year sequence screening threshold similar to NUREG-1855, Chapter 9. In the context of this methodology, the parametric uncertainty evaluation for non-seismic sequences includes the following steps:

- The TR specifies if the best estimate mean sequence frequency is below 1E-8 per module year, more than an order of magnitude below the screening threshold, then the sequence is screened out from the EPZ technical basis and removed from further consideration.

- The TR specifies if the best estimate mean sequence frequency is within an order of magnitude, then then 95th percentile value will be compared against the 1E-7 per module year numerical screening threshold.
    - If the 95th percentile value is below the 1E-7 per module year screening threshold, then the sequence is also screened out from the EPZ technical basis and removed from further consideration.
    - If the 95th percentile value is above the 1E-7 per module year screening threshold, then the sequence is screened into the process and included in the source term and dose analysis.

- The TR specifies that if the best estimate mean sequency frequency is above 1E-7 per module year, then the sequence is screened into the process and included in the source term and dose analysis.

Based on its review, the NRC staff finds that the approach for addressing the uncertainty in the numerical screening threshold for all hazards and all modes except seismic hazard is acceptable and will capture a significant portion of the corresponding parametric uncertainty while being consistent with Chapter 9 of NUREG-1855, Revision 1.

<u>Uncertainty in the Numerical Screening Threshold for Seismic Hazard</u>

As noted above, Section 5, Stage C-1.2 of NUREG-1855 identifies {{ }} for use when screening is performed based on {{ }} Further, the NRC staff notes that because {{ }} is derived from the underlying distribution, {{ }} includes a portion of the parametric uncertainty in {{ }} and is in the higher percentages of the distribution.

{{

}}

The NRC staff's review determined that the identification and inclusion of new sequences {{ }} provide appropriate consideration of uncertainty in the seismic event screening threshold and address potential {{

}}

Modeling Uncertainty in PRAs

The NRC staff recognizes that a user of this TR will develop technically acceptable all hazards, all modes PRAs. These PRAs may be developed prior to construction of the as-built plant, and will have no operating data, contributing to PRA uncertainty. As discussed in RG 1.200, Revision 3, key assumptions and sources of uncertainty for a PRA application are generally identified from the assumptions and approximations identified in the base PRA. These key assumptions are used to identify sensitivity studies as input to the decision-making associated with the application. NUREG-1855 provides detailed guidance on identification and disposition of key assumptions and sources of uncertainty in the context of the risk-informed application.

NuScale's TR discusses key assumptions related to PRA modeling in Section 3.8, "Reviewing Key Uncertainties and Sources of Uncertainty in the Underlying PRA." Section 3.8 explains that the user of the TR needs to complete a review of the assumptions and sources of uncertainty in the underlying PRA to identify and address any potential impact(s) on the application of the methodology. Based on its review, the NRC staff determined the need for a "Condition of Use" that would result in consistency with NRC guidance on the identification and disposition of key assumptions and sources of uncertainty for the PRAs used in this methodology. As stipulated in Section 5.0 of this SE (Condition C), the user of this TR will provide a discussion of: (1) how PRA key assumptions and sources of uncertainty for each hazard and mode were identified for this application, and (2) how the impact of each identified key assumption and source of uncertainty on the decisions using the methodology, including the numerical screening thresholds, was assessed and dispositioned. The identification, assessment, and dispositioning of key assumptions and sources of uncertainty will be consistent with the guidance in RG 1.200, Revision 3 and NUREG-1855, Revision 1.

Summary of NRC Staff's Review of the Use of PRA and PRA Insights in the Methodology

For EPZ sizing considerations, the NRC staff reviewed the screening criteria for non-seismic initiating events and seismic initiating events evaluated in the PRA as specified in Revision 3 of the TR. The NRC staff found the screening criteria to be consistent with NUREG-0396 to identify a spectrum of accidents for which the potential consequences, timing, and release characteristic will be determined.

In summary, the NRC staff finds that: (1) the methodology in the TR uses PRAs and insights from those PRAs consistent with the Commission's PRA Policy Statement, and risk-informed decision making discussed in SRM to SECY-98-144 and RG 1.174, Revision 3, (2) the use of the TR, in conjunction with the NRC staff's conditions of use, is supported by PRAs that follow NRC guidance for demonstrating their technical acceptability, (3) the use of the TR, in conjunction with the NRC staff's Conditions of Use, appropriately addresses uncertainty, including key assumptions and sources of uncertainty, consistent with relevant NRC guidance, and (4) the screening criteria for non-seismic initiating events and seismic initiating events, including consideration of uncertainty and cliff-edge effects, for EPZ sizing as specified in the TR, are consistent with NUREG-0396 to identify a spectrum of accidents for which the potential consequences, timing, and release characteristic will be determined.

3.7 Evaluation of Performance Measurement Strategies

The NRC staff evaluated the methodology in the TR to determine whether it included performance measurement strategies consistent with the principles of risk-informed decision making in RG 1.174, Revision 3, and the Commission's expectations in the SRM to SECY-98-0144.

Based on its review, the NRC staff determined that a "Condition of Use" on this TR was appropriate to ensure that the inputs and decisions made using the TR's risk-informed methodology continue to remain valid as the plant and the supporting PRAs are updated. The objective of this periodic evaluation is to ensure that the condition of applicability on the plant-level fragility (Condition E in Section 5.0 of this SE) is met and that conclusions about the EPZ size made using the TR's methodology remain unchanged. Therefore, as stipulated in Section 5.0 of this SE, the user of this TR will propose a condition on its COL or operating license for a periodic evaluation that determines whether: (1) item E of the "Conditions of Use," as described above in this section of the SE is met, and (2) conclusions about the EPZ size using the TR methodology remain unchanged. The periodic evaluation in the proposed condition will be performed following an update to the user's PRAs based on a review of changes to the plant structures, systems, and components, operational practices, and applicable plant and industry operational experience. The periodicity of the evaluation in the proposed condition will be consistent with 10 CFR 50.71(h)(2). Prior NRC approval under 10 CFR 50.90 will be required for any changes to the EPZ size.

3.8 Evaluation of Consistency with Risk-Informed Decision-making

As described above, the NRC staff has found Revision 3 of the TR to be consistent with risk-informed decision making defined in SRM to SECY 98-144 and the principles of risk-informed decision making in RG 1.174, Revision 3. The TR adequately addresses the use of risk insights from a technically acceptable PRA and adequately addresses other factors such as defense-in-depth and the identification and evaluation of key sources of uncertainty. The NRC staff determined that a "Condition of Use" on this TR was appropriate to ensure that the inputs and decisions made using the risk-informed methodology continue to remain valid as the plant and the supporting PRAs are updated.

4.0 NRC Staff Conclusion

The NRC staff has completed its review of TR0915-17772, Revision 3, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale SMR Plant Sites." Based on the results of the NRC staff's technical evaluation documented in Section 3.0 of this SE, the NRC staff concludes there is reasonable assurance that the proposed methodology is adequate for assessing plume exposure pathway EPZ sizing for the scope of applicability provided in Section 2.5 of the TR. Therefore, subject to the scope of applicability provided in Section 2.5 of the TR and the Conditions of Use listed in Section 5.0 of this SE, a licensee or an applicant may use the TR as the technical basis for assessing plume exposure pathway EPZ sizing.

5.0 Conditions of Use

The use of this TR is subject to the following conditions of use:

A. Use of this TR by an applicant for an operating license under 10 CFR Part 50 is permitted if:

    1. The applicant submits the design- and site-specific PRA results and insights for NRC staff review as part of their Final Safety Analysis Report, consistent with the

Commission expectations for the use of the PRA, as described in the Statement of Considerations for 10 CFR Part 52 (72 FR 49387), and

   2. The level of design detail in the design- and site-specific PRAs is commensurate with a 10 CFR Part 52 combined license (COL) application.

B. The requirements for PRA technical acceptability for the PRAs used for the methodology in this TR are as follows:

   1. The PRAs used for this TR will be design- and site-specific, and will be developed for all modes and hazards, including seismic events, using the guidance in RG 1.200, Revision 3, at Capability Category II. Any exceptions to meeting Capability Category II (e.g., inability to perform walkdowns) must be identified and justified by the user of the TR in its application.

   2. The PRAs used for this TR will be peer reviewed in accordance with the guidance in RG 1.200, Revision 3.

   3. The PRAs used for this TR will address hazards and modes not covered by endorsed consensus PRA Standards, and the user will justify the technical acceptability of these PRAs for use in identifying an acceptable spectrum of events.

C. The user of this TR will provide a discussion of (1) how PRA key assumptions and sources of uncertainty for each hazard and mode were identified for this application, and (2) how the impact of each identified key assumption and source of uncertainty on the decisions using the methodology, including the numerical screening thresholds, was assessed and dispositioned. The identification, assessment, and dispositioning of key assumptions and sources of uncertainty will be consistent with the guidance in RG 1.200, Revision 3 and NUREG-1855, Revision 1.

D. {{
}} in this TR is allowed only if the site-specific probabilistic GMRS is bounded by NuScale's CSDRS for the horizontal direction provided in Table 3.7.1-1 of the NuScale Final Safety Analysis Report (ML20224A484). The GMRS used to demonstrate that this condition is met will be derived using the approach in RG 1.208, Revision 0 (ML070310619) and the site-specific probabilistic seismic hazard analysis accepted by the NRC.

E. Use of {{
}} in this TR is allowed only if the high confidence of low probability of failure (HCLPF) plant-level fragility is equal to or greater than 0.84g at peak ground acceleration (100 Hertz). The HCLPF plant-level fragility will be the peak ground acceleration at the probability of 0.01 on the mean conditional core damage probability curve derived from a seismic PRA that meets item B of the conditions of use, as described above in this section of the SE.

F. A user of this TR will demonstrate that items D and E of the conditions of use, as described above in this section of this SE, are met at the time of submission of an application using this methodology and prior to fuel load. The demonstration prior to fuel load will be based on the as-built plant.

G. {{ }} of this TR will be monotonically decreasing and demonstrate that the probability of large doses drops off substantially.

H. The user of this TR will propose a condition on its construction and operating license (COL) or operating license for a periodic evaluation that ensures: (1) item E of the conditions of use, as described above in this section of the SE, is met, and (2) conclusions about the EPZ size using the TR methodology remain unchanged. The periodic evaluation in the proposed condition will be performed following an update to the user's PRAs based on a review of changes to the plant structures, systems, and components, operational practices, and applicable plant and industry operational experience. The periodicity of the evaluation in the proposed condition will be consistent with 10 CFR 50.71(h)(2). Prior NRC approval under 10 CFR 50.90 will be required for any changes to the EPZ size resulting from an inability to meet the proposed condition.

6.0 References

1. NuScale Power, LLC, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," Licensing Topical Report TR-0915-17772, Revision 3, June 2022 (ADAMS Accession No. ML22161B011 [proprietary]; ML22161B010 [non-proprietary]).

2. NuScale Power, LLC, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zone," Licensing Topical Report TR-0915-17772, Revision 2 (ADAMS Accession No. ML20217L422).

3. NuScale Power, LLC "NuScale Response to NRC Request for Additional Information eRAI No. 9828," RAIO-120112 (ADAMS Accession No. ML22161A995 [proprietary]; ML22161A994 [non-proprietary]).

4. NuScale Power, LLC "Response to NRC Request for Additional Information No. 9830 (eRAI 9830) on the NuScale Topical Report, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones, TR-0915-17772, Revision 2," RAIO-103182, May 26, 2021 (ADAMS Accession No. ML21146A363)

5. Code of Federal Regulations, *Title 10, Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."

6. Code of Federal Regulations, *Title 10, Energy*, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

7. United States Nuclear Regulatory Commission, "Licenses, Certifications, and Approvals for Nuclear Power Plants," *Federal Register* 72 FR 49351, August 28, 2007.

8. Code of Federal Regulations, *Title 10, Energy*, Section 50.47, "Emergency Plans."

9. Code of Federal Regulations, *Title 10, Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

10. Code of Federal Regulations, *Title 10, Energy*, Part 20, “Standards for Protection Against Radiation.”

11. Code of Federal Regulations, *Title 10, Energy*, Section 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report.”

12. United States Nuclear Regulatory Commission, “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants,” NUREG-0396, December 1978 (ADAMS Accession No. ML051390356).

13. United States Nuclear Regulatory Commission, “Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents,” *Federal Register* 44 FR 61123, October 23, 1979.

14. United States Nuclear Regulatory Commission, “Severe Reactor Accidents Regarding Future Designs and Existing Plants,” *Federal Register* 50 FR 32138, August 8, 1985.

15. United States Nuclear Regulatory Commission, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory,” *Federal Register* 60 FR 42622, August 16, 1995.

16. United States Nuclear Regulatory Commission, “Staff Requirements – SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationships to Current Regulatory Requirements,” SRM-SECY-90-016, June 26, 1990 (ADAMS Accession No. ML003707885).

17. United States Nuclear Regulatory Commission, “Staff Requirements – SECY-04-0118 - Plan for the Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality,” SRM-SECY-04-0118, October 6, 2004 (ADAMS Accession No. ML042800369).

18. United States Nuclear Regulatory Commission, “Staff Requirements – SECY-98-144 - White Paper on Risk-Informed and Performance-Based Regulation,” SRM-SECY-98-144, October 6, 2004 (ADAMS Accession No. ML003753601).

19. United States Environmental Protection Agency, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,” EPA-520/1-75-001, September 1975.

20. United States Environmental Protection Agency, “Manual Protective Action Guides and Planning Guidance for Radiological Incidents,” EPA-400/R-17/001, January 2017.

21. United States Nuclear Regulatory Commission, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, Revision 3, January 2018 (ADAMS Accession No. ML17317A256).

22. United States Nuclear Regulatory Commission, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Regulatory Guide 1.200, Revision 3, December 2020 (ADAMS Accession No. ML20238B871).

23. United States Nuclear Regulatory Commission, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” NUREG-1855, Revision 1, March 2017 (ADAMS Accession No. ML17062A466).

24. United States Nuclear Regulatory Commission, “Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application,” Interim Staff Guidance, DC/COL-ISG-028, November 2016 (ADAMS Accession No. ML16130A468).

25. United States Nuclear Regulatory Commission, “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” WASH-1400 (also NUREG 75/014), October 1975.

26. NuScale Power, LLC, “NuScale Standard Plant Design Certification Application Chapter Nineteen Probabilistic Risk Assessment and Severe Accident Evaluation,” Revision 5, July 2020 (ADAMS Accession Number ML20224A508).

27. United States Nuclear Regulatory Commission, “Audit Summary for the Regulatory Audit of NuScale Power, LLC, Topical Report TR-0915-17772, “Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones, Revision 2,”” June 14, 2022 (ADAMS Accession No. ML22146A399).

28. United States Nuclear Regulatory Commission, “Assessment of Dosimetry Data for Accidental Radionuclide Releases from Nuclear Reactors,” NUREG/CR-4185, September 1985 (ADAMS Accession No. ML20132F001)

29. United States Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150 (available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/index.html)

30. United States Nuclear Regulatory Commission, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” NUREG-1935 (available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1935/index.html)

31. United States Nuclear Regulatory Commission, “Health Effects Models for Nuclear Power-Plant Accident Consequence Analysis Part I: Introduction, Integration, and Summary,” NUREG/CR-4124, Rev. 2, Part I, October 1993 (ADAMS Accession No. ML050030192).

32. United States Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Transient and Accident Analysis,” NUREG-0800, Chapter 15 (available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/ch15/index.html)

33. United States Nuclear Regulatory Commission, “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” WASH-

1400 (also NUREG 75/014), Appendix VI, October 1975 (ADAMS Accession Number ML070600389).

34. Atkinson, D., and Lee, R.F., “Procedures for Substituting Values for Missing NWS [National Weather Service] Meteorological Data for Use in Regulatory Air Quality Models,” July 1992 (available at https://www.epa.gov/sites/default/files/2020-10/missdata.txt)

35. United States Nuclear Regulatory Commission, “Use of MACCS Dose Coefficient Files to Compute Total Effective Dose Equivalent,” July 30, 2021 (ADAMS Accession No. ML21211A584).

36. United States Nuclear Regulatory Commission, “Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions,” NUREG/CR-4467, March 1986 (ADAMS Accession No. ML093490126).

37. United States Nuclear Regulatory Commission, “MELCOR Accident Consequence Code System,” NUREG/CR-4691, Vol. 2, February 1990 (ADAMS Accession No. ML063560409).

38. United States Nuclear Regulatory Commission, “MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project,” August 2014 (ADAMS Accession No. ML14234A148).

39. United States Environmental Protection Agency, “Workbook of Atmospheric Dispersion Estimates,” 1970 (available via https://nepis.epa.gov).

40. United States Nuclear Regulatory Commission, “SOARCA Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station”, NUREG/CR-7155, May 2016 (ADAMS Accession No. ML16133A461).

41. International Atomic Energy Agency, “Defense in Depth in Nuclear Safety,” INSAG-10, 1996 (available at https://www.iaea.org/publications).

42. United States Nuclear Regulatory Commission, “Emergency Planning,” *Federal Register* 45 FR 55402, August 19, 1980.

43. United States Nuclear Regulatory Commission, “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication” *Federal Register* 51 FR 30028, August 21, 1986.